NON-PUBLIC?: N
ACCESSION #: 8712080038

LICENSEE EVENT REPORT (LER)

FACILITY NAME: Vermont Yankee Power Station PAGE: 1 of 5

DOCKET NUMBER: 05000271

TITLE: Main Turbine Trip And Reactor Scram From Feedwater Valve Malfunction Due To Personnel Valve Repair Error
EVENT DATE: 11/08/87 LER #: 87-017-00 REPORT DATE: 12/04/87

OPERATING MODE: N POWER LEVEL: 088

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: JAMES P. PELLETIER, PLANT MANAGER TELEPHONE #: 802-257-7711

COMPONENT FAILURE DESCRIPTION:
CAUSE: B SYSTEM: SJ COMPONENT: FCV MANUFACTURER: F130
REPORTABLE TO NPRDS: Y

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT: On 11-8-87, at 0124, during a plant power reduction from 100% to 80% in preparation for surveillance testing, the two Feedwater Regulating valves (EIIS=FCV) failed to throttle closed as required by decreased Feedwater flow demand. This was due to a mechanical restriction of the manual operators on these valves that was added approximately three weeks earlier to temporarily repair the operators. When installed, it was not recognized that future full stroke valve motion would be restricted by the repair.

The failure of these valves to close as required created a high Reactor Water Level and subsequent Turbine trip and Reactor SCRAM at 0126. All automatic actions occurred as required. All systems were stabilized and returned to normal within two minutes.

An unexpected Primary Containment Isolation System (PCIS) actuation of the Main Steam Isolation Valves (MSIV's) occurred during operator SCRAM response as a result of the Reactor Mode Switch being moved from RUN. This switch movement was done prior to Main Steam flow instrumentation resetting from decreased steam flow from the SCRAM. PCIS isolations were promptly reset.

Valve operator restrictions were removed and repaired and full stroke of the valves was verified. Plant personnel will be further trained in the proper operation of these valves. A procedure change has been made to preclude the inadvertent Reactor Mode Switch initiated PCIS isolation.

(End of Abstract)

DESCRIPTION OF EVENT

On 11-8-87, reactor power was being reduced using Recirculating Pump speed control from 100% to 80% power to allow for Main Turbine Bypass Valve routine surveillance testing. The Feedwater system responded to this power change as expected by slowly closing the two Feedwater Regulating Valves as Reactor steam output diminished.

At 0124, with reactor power at 90%, both Feedwater Regulating Valves (Fisher Controls Model #476L-5-HSV) (EIIS = FCV) malfunctioned and would not close any further. This created a mismatch between reactor steam flow and feed-water flow, causing Reactor Water Level to rise.

Upon observing the water level rise, operators entered the Reactor High Level transient procedure. Per this procedure, the Feedwater Regulating valves were placed from automatic to manual control. However, they could still not be closed further, and Reactor Water Level continued to rise until a high level Main Turbine trip and subsequent reactor SCRAM occurred at 88% power at 0126.

After receiving the automatic SCRAM, operators entered the SCRAM Procedure. Per this procedure, the SCRAM was visually verified (all rods inserted) and Reactor Power was verified to be less than 2%. At this point, the procedure requires operators to verify that steam flow in each Main Steam line is less than 40% and then transfer the Reactor Mode Switch from the RUN position to the SHUTDOWN or REFUEL position. (Note: If the Reactor Mode Switch is moved from the RUN position at Main Steam flows of 40% or greater, a "High Steam Flow in SHUTDOWN" isolation of the Main Steam Isolation Valves (MSIV's) occurs as part of the Primary Containment Isolation System (PCIS) and is designated a PCIS Group 1 isolation).

After verifying steam flow to be less than 40%, operators transferred the Reactor Mode Switch out of RUN and still received the above discussed PCIS Group 1 isolation of the MSIV's. This isolation created a Reactor pressure increase and corresponding water level decrease which initiated PCIS Groups 2, 3, and 5 due to the resulting low Reactor Water Level. (Note: These PCIS

Groups isolate Primary Containment penetrations and the Reactor Water Cleanup System.)

Reactor Water Level returned to its normal level in 12 seconds and continued to increase until the two operating Reactor Feedwater Pumps tripped on high Reactor Water Level 57 seconds after the event. After clearing the PCIS Group 1 logic, the MSIV's were reopened within two minutes of the event. In summary, all systems were stabilized within two minutes of the event.

CAUSE OF EVENT

On 10-19-87, approximately three weeks prior to the subject event, plant operators observed a mismatch in the positions of the "A" and "B" Feedwater Regulating Valves. Since these valves are in parallel and normally in the same position, a technician was sent to investigate.

CAUSE OF EVENT (Cont.)

It was found that a threaded sleeve surrounding the valve stem that is used for manual operation of the valve had rotated from its normal "neutral" position down the stem and was restricting movement of the "B" valve. (Note: Sleeve rotation is normally prevented by a key that engages the sleeve along an axial slot). The "A" valve continued to open to maintain Reactor Vessel level, thus creating the above mismatch in valve position.

The threaded sleeve on the "B" valve was raised back up to stem and wired to the valve yoke to preclude it rotating down again. The "A" valve was also wired to prevent a similar occurrence although it was fully operable. However, both valves were wired in a position other than "neutral". Since the neutral position of the sleeve is required for full valve stroke, the mispositioning of the sleeve restricted valve closure, causing high water level on 11-8-87, which is the immediate cause of this event.

The intermediate cause has been determined to be the broken/missing key which is designed to keep the collar from rotating and moving vertically. Since the "B" valve in question has not been disassembled for 22 months and has operated satisfactorily during that period, it is assumed that this key may have recently become dislodged/broken rather than its being not installed in the valve. An additional intermediate cause is the lack of documentation and review of the wiring configuration using the formal Maintenance Request process. Although a Maintenance Request (MR) existed for a handwheel problem related to the broken/missing key on the "B" valve, the scope of repair was not revised to document the installation of wire on the valve operator.

The evaluation of this event has identified the root cause to be personnel unfamiliarity with proper operation of these valves, thus resulting in restricted valve motion from the temporary repair (wiring the threaded sleeve out of position) that was applied to the "B" valve, and unnecessary modification of the operable "A" valve.

The immediate cause of the PCIS Group 1 isolation resulted from the Reactor Mode Switch being moved out of run prior to the flow switches which initiate the isolation reaching their reset point. These switches are "armed" during Reactor startup upon reaching a steam flow corresponding to 40% power, increasing. However, due to hysteresis, the switches do not "disarm" at 40% power, decreasing, but at a slightly lower setpoint.

Since the SCRAM procedure being followed directs operators to move the Mode Switch after verifying steam flow is less than 40%, the subject PCIS isolation can occur if the Mode Switch is moved while switches are in this dead band region just below 40%, even if the procedure is followed.

The root cause of the PCIS Group 1 isolation is therefore attributed to a procedural deficiency in not recognizing this dead band region of the flow switches on decreasing steam flow.

ANALYSIS OF EVENT

At all times during the event, the Feedwater system (a Non-nuclear safety system) was capable of producing 100% of its designed flow. In addition, during the events discussed:

a) The Reactor Protection System (RPS) responded as designed.

b) The PCIS responded as designed.

c) Operators responded as required by procedure.

d) Technical Specifications were satisfied at all times during the event.

e) Emergency Core Cooling Systems (ECCS) were operable during the event to provide Reactor Vessel inventory control, although they were not initiated.

Therefore, during the events of this report, there were no adverse safety implications to plant equipment, personnel, or to the public.

CORRECTIVE ACTIONS

Specific training concerning the operation of these valves will be given to all affected personnel and the importance of Maintenance Request documentation will be stressed.

A procedure change, which was being processed at the time of the event, has been made to the Feedwater procedure to assure that the threaded sleeve around the valve stem is returned to the neutral position after manual operation to assure full valve stroke capability.

A bolted clamp has been added to the sleeve of the "B" valve to prevent it from turning. The "B" valve has been verified to stroke fully with this modification. The "A" valve has been returned to its original unaltered configuration and also full stroke tested.

During the next refueling outage, the "B" valve operator will be disassembled and inspected for a missing/broken key to determine the exact failure. In addition, the sleeves on both valves will be checked in the neutral position on daily operator rounds to assure their continued full stroke operability during the current operating cycle. To prevent inadvertent repositioning of the threaded sleeve, lead seals have been added to both valve handwheels to prevent their operation.

To preclude inadvertent PCIS Group 1 isolations following Reactor SCRAMS, the SCRAM procedure has been revised to assure operators are directed to move the Mode Switch after steam flow switches have reset. This is at a steam flow corresponding to less than 40% power.

ADDITIONAL INFORMATION

A Nuclear Plant Reliability Data System (NPRDS) inquiry showed no evidence of similar valve problems at other plants.

No similar occurrences have been reported to the Commission in the last five years.

ATTACHMENT # 1 TO ANO # 8712080038 PAGE: 1 of 1

VERMONT YANKEE NUCLEAR POWER CORPORATION

P.O. BOX 157
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VERNON, VERMONT 05354

December 4, 1987
VYV-87-270

U.S. Nuclear Regulatory Commission
Document No. 50-271
Washington, D.C. 20555

REFERENCE: Operating License DPR-28
Docket No. 50-271
Reportable Occurence No. LER 87-17

Dear Sirs:

As defined by 10CFR50.73, we are reporting the attached Reportable Occurrence as LER 87-17.

Very truly yours,

VERMONT YANKEE NUCLEAR
POWER CORPORATION

/s/James P. Pelletier

Plant Manager

cc: Regional Administrator
USNRC Office of Inspection and Enforcement
Region I
631 Park Avenue
King of Prussia, Pennsylvania 19406

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